Supplementary Shielding Calculation for the GNS 11 Transport Cask Loaded with 33 Square-shaped MTR Fuel Assemblies of the Type MERLIN

GNS

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Supplementary Shielding Calculation
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0. Summary

The GNS 11 transport cask is designed to transport MTR fuel assemblies from research reactors (e.g. GKSS, FRJ, FRG etc.).

Both square-shaped fuel assemblies (MERLIN) and tubular fuel assemblies (DIDO) can be transported in different basket variations.

In the GNS report B 37/86, "Safety Analysis Report, GNS 11", chapter 5 it has already been demonstrated that the allowable limits of the transport regulations (2000 µSv/h on the surface and 100 µSv/h at a distance of 2 m) are observed for loading variations with 28 DIDO and 33 MERLIN fuel assemblies. The fuel assemblies concerned were, among other things, highly enriched (HEU) box-shaped fuel assemblies with the following specification:

- 93 % enrichment (≙ 264 g U-235 / F/A with 330 g uranium / F/A)
- Burn-up: approx. 50 % U-235
- Decay time: 180 days

To supplement chapter 5 of the safety report GNS B 37/86, the present report demonstrates that observance of the above limits is also ensured when the cask is loaded with 33 square-shaped fuel assemblies, for example of the type MERLIN, with LEU - (low enrichment) fuel. The reference LEU fuel assembly taken here has the following specification:

- Content of fissile material (fresh): 323 g U-235 / F/A with 1635 g total uranium / F/A (≙ 19.75 % enrichment)
- Burn-up (mean): 65 % U-235 (≙ 178 MWd / F/A)
- Decay time: 360 days

The results of selected points on the GNS 11 show that the above limits from the transport regulations are observed.

MTR fuel assemblies with medium initial enrichment (MEU) and comparable burn-up are covered by the safety framework formed by the LEU and HEU F/As with regard to shielding.

1. Introduction and Problem

The GNS 11 is used as a transport cask to dispose of spent tubular MTR fuel assemblies (DIDO) and square-shaped MTR fuel elements (MERLIN) from German research reactors.

The GNS safety report B 37/86 was concerned with the transport of 28 DIDO and 33 MERLIN fuel assemblies with different degrees of enrichment, and it provided evidence that the allowable dose rate limits laid down in the transport regulations are observed by the GNS 11.

This report - which supplements chapter 5 of the aforementioned safety report - checks that with loads of 33 square-shaped MERLIN fuel asssemblies with LEU (low enirchment) fuel, the limits of the transport regulations are again observed.

The reference fuel assembly was one with 23 fuel plates which is used in the same design in German reactors (GKSS, FRM, DER and FRG) and in other research reactors (e.g. FZS, PSI etc.).

The calculations were based on reference fuel assemblies with the following specification.

- Content of fissile material (fresh): 323 g U-235 / F/A with 1635 g total uranium / F/A ($\hat{=}$ 19.75 % enrichment)
- Burn-up (mean): 65 % U-235 ($\hat{=}$ 178 MWd / F/A)
- Decay time: 360 days

With a view to imminent or future transport operations the above fuel assembly specification also covers other fuel assemblies with lower burn-ups but with the same decay time.

2. Specification of the Fuel Assemblies

2.1 General

The structure of the MTR reference fuel assemblies used here (box-shaped with 23 fuel plates per fuel assembly) was taken from the safety report GNS B 37/86.

The fissile material was considered as U_3O_8 / Al dispersion. Because of the additional (α, n) reactions with the oxygen, this composition for the fissile material is a design determinant with regard to the neutron emission rate of the spent fuel assemblies as compared with the fissile material compositions U_3Si_2 or UAl_x.

With regard to the content of fissile material, these fuel assemblies also cover tubular MTR fuel assemblies or box-shaped MTR fuel assemblies with a small number of fuel plates per fuel assembly. The caluclations described below are based on calculated activity inventories, obtained with the help of the burn-up computer program ORIGEN -2 /1/. The burn-up calulations were conducted under the following parameters:

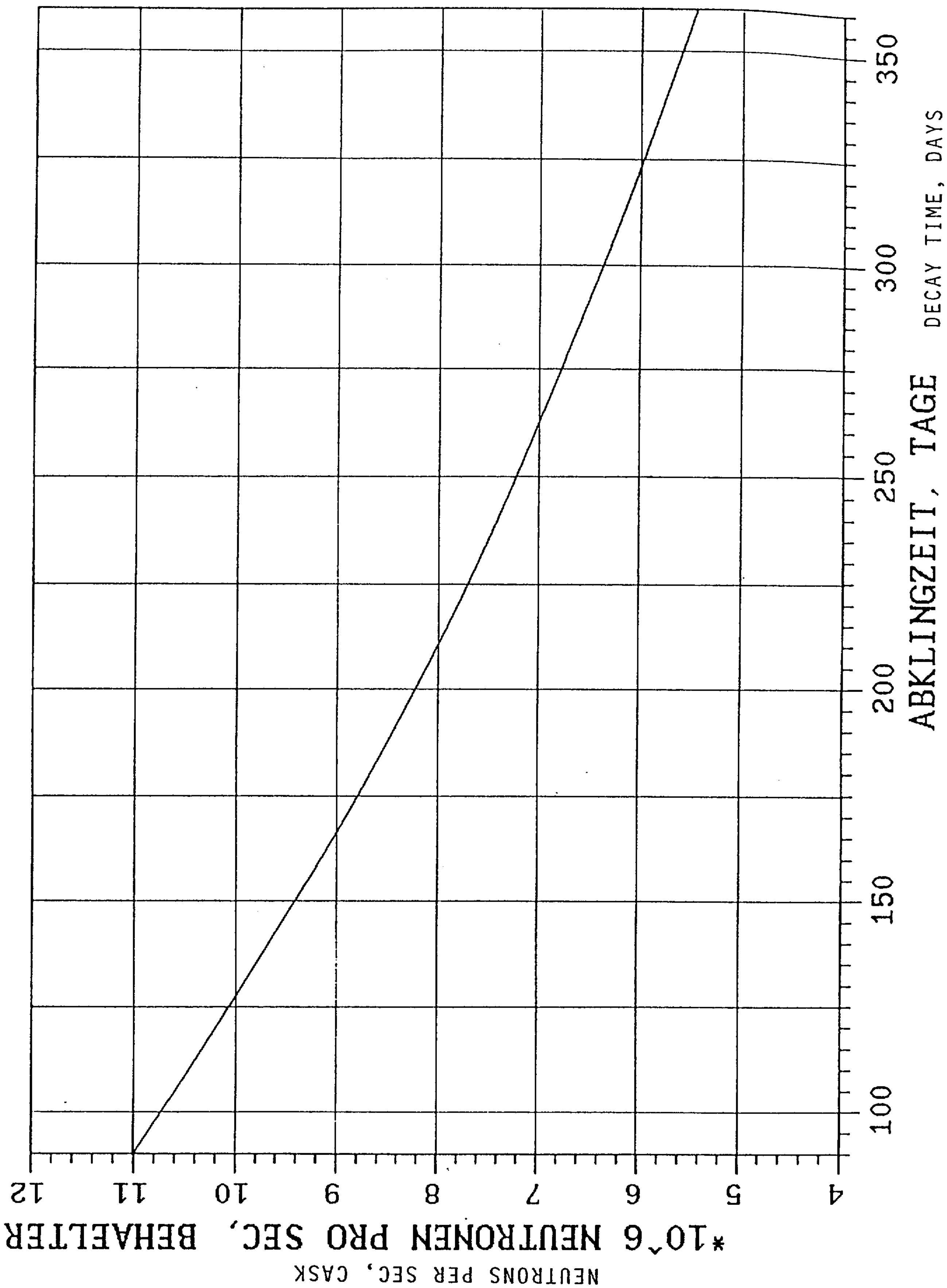
- Service life:	828 d
- Target burn-up (mean):	65 % U-235 (≙ 178 MWd/ F/A)
- F/A capacity:	214.96 KW /F/A
- Fissile material content (fresh):	323 g U-235 (≙ 19.75 % enrichment)

2.2 Emission Rates

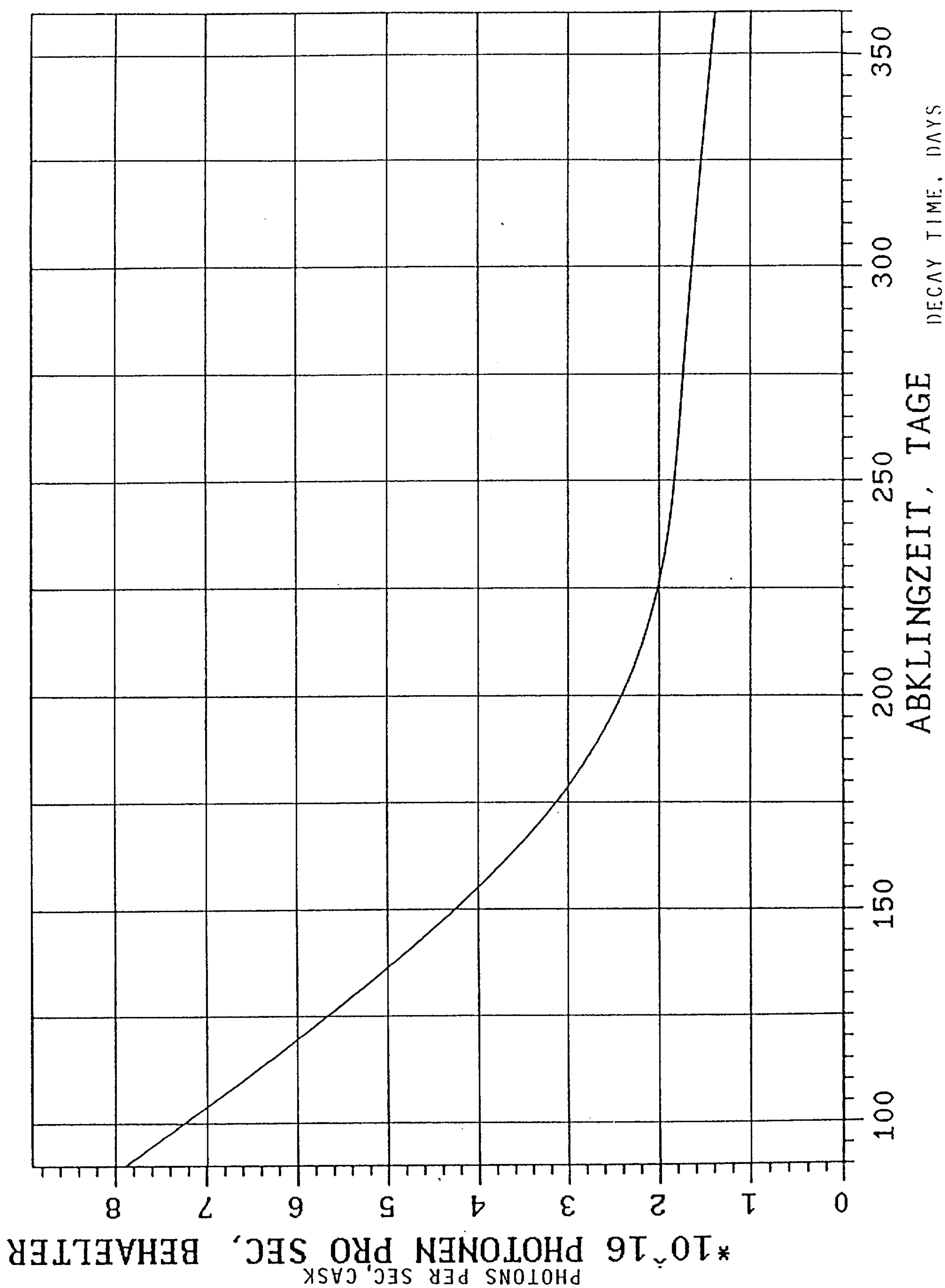
The neutron and photon emission rates resulting from the burn-up calculations, as well as the decay after-heat rate are plotted in Figs. 1 to 3 for the maximum cask inventory (33 MTR F/As) as a function of the decay time.

Table 1 shows the photon spectrum and neutron emission rate for 360 days decay time. As can be seen in Fig. 3, the maximum removable heat rate of approx. 1.6 KW / cask is only reached for this inventory after more than one year decay time.

The source terms for a decay time of 360 days therefore cover conservatively the MTR LEU fuel assemblies under consideration.



GNS	Neutron Emission as a function of the Decay Time for a load of 33 LEU Fuel Assemblies	Fig. 1



GNS	Photon Emission as a Function of the Decay Time for a Load of 33 LEU Fuel Assemblies	Fig. 2

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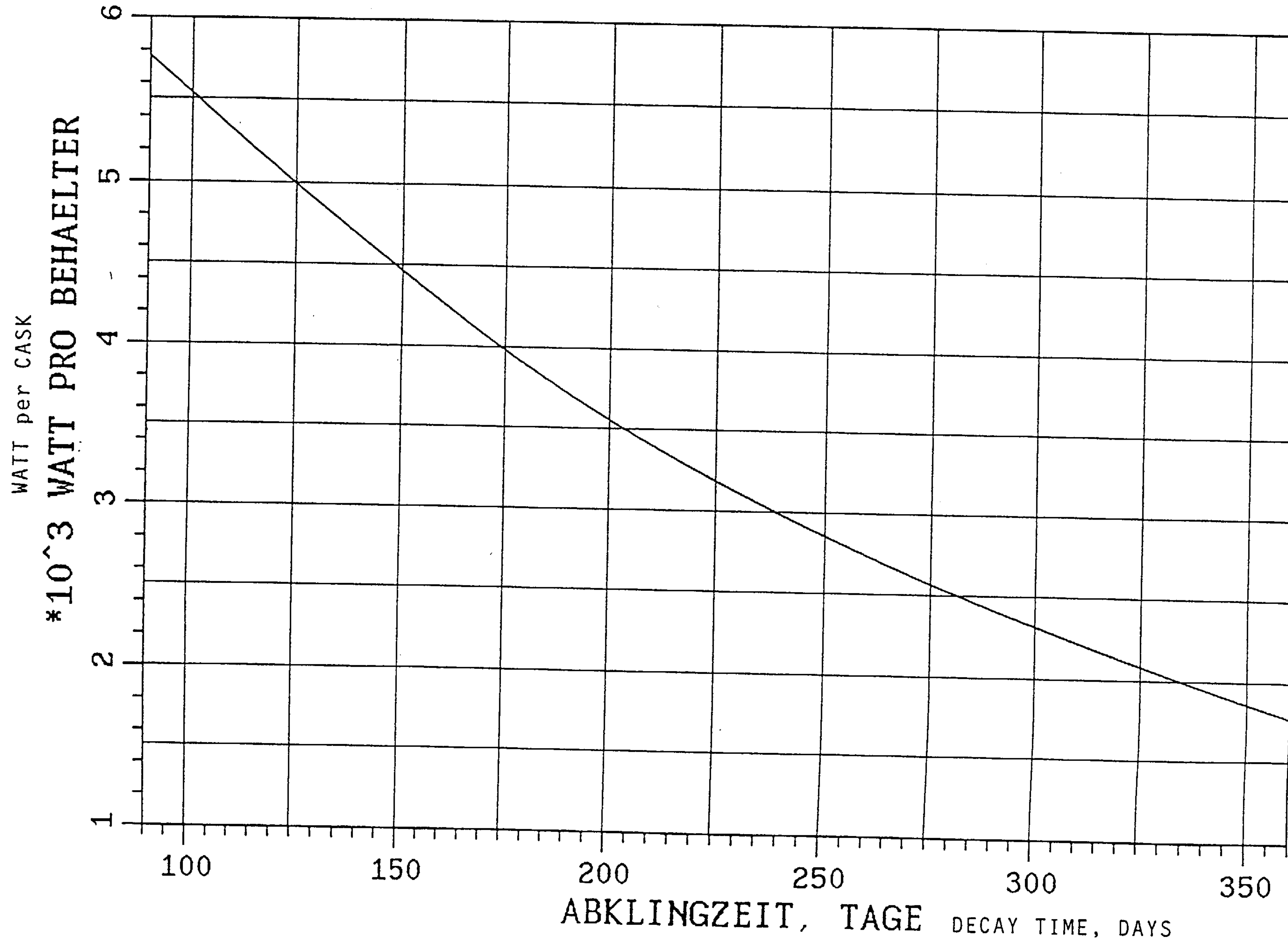
Decay After-Heat Rate as a Function of the Decay Time for a Load of 33 LEU Fuel Assemblies

Fig. 3



Photon Emission

Gamma energy MeV	γ emission rate s^{-1} / F/A	γ emission rate s^{-1} / cask
0.375	9.03 E 12	2.98 E 14
0.575	5.88 E 13	1.94 E 15
0.850	4.18 E 13	1.38 E 15
1.25	3.48 E 12	1.15 E 14
1.75	2.49 E 11	8.22 E 12
2.25	9.73 E 11	3.21 E 13
2.75	5.03 E 9	1.66 E 11
3.50	5.49 E 8	1.81 E 10
Total	1.14 E 14	3.77 E 15

Neutron Emission

α-n neutrons	5.61E4 n/s.F/A	1.85E6 n/s.cask
Neutrons from spontaneous fission	1.09E5 n/s.F/A	3.60E6 n/s.cask
Total	1.65E5 n/s.F/A	5.45E6 n/s.cask

Enrichment: 19.75 % U-235
F/A burn-up 65 % U-235 ($\hat{=}$ 178 MWD / F/A) (mean)
Decay time: 360 days

Photon Spectrum and Neutron Emission
Rate with 360 Days Decay Time Table 1

3. Description of the Shielding Geometry

For the shielding calculations conducted on the GNS 11, not only the shielding geometry, but also the corresponding shielding materials and material densities were taken from the safety report for the GNS 11.

Table 2 shows the materials, densities and wall thicknesses taken into account for the shielding calculations in the shell, lid and bottom areas of the cask.

	Material	Density g/cm^3	Wall thickness cm
Source	Iron Aluminium Uranium	2.06*	33.0
Shell	Air	0.0012	3.15
	Iron (liner)	7.9	2.0
	Lead	11.3	16.5
	Iron liner	7.9	4.0
Lid	F/A heads/basket	1.8	6.35
	Air	0.0012	6.3
	Iron (lid)	7.7	26.5
	Iron (protective plate)	7.9	3.5
	Iron (shock abs.)	7.85	0.5
	Wood (shock abs.)	0.7	14.0
	Iron (shock abs.)	7.85	0.5
Bottom	F/A feet/basket	1.66	21.25
	Iron (liner)	7.9	4.0
	Lead	11.3	13.8
	Iron (liner)	7.9	4.0
	Iron (shock abs.)	7.85	0.5
	Wood (shock abs.)	0.7	14.0
	Iron (shock abs.)	7.85	0.5

*Basket material and F/A material assumed as distributed homogeneously in a cylindrical source.

Table 2: Materials, Densities and Wall Thicknesses for the Shielding Calculations on the GNS 11

4. Computational Method

4.1 Computer Model for Gamma Shielding

The calculation of the gamma shielding is conducted with a computer program based on the "POINT-CORE METHOD" /2,3/.

With the help of the point-kernel method it is possible to conduct three-dimensional caluclations with multilayer shielding materials, taking due account of the energy of the emitted photons.

The computer program "ZYLIND" /4/ is used to determine the gamma dose rates.

4.2 Computer Model for Neutron Shielding

The shielding calculations of the neutron source are based on the "ANISN" computer code (a one-dimensional discrete coordinates transport code with anisotropic scattering) /5/.

ANISN resolves the one-dimensional, energy-related Boltzmann transport equation with anisotropic scatter of higher order in plate, ball or cylinder geometry for neutrons and photons. Eigen value problems can be dealt with and fixed sources can be given. A fission source, an external source (volumetric or area-type source) or a combination of both can be given for subcritical cases. For various variables such as concentration, outside radius, buckling, zone thickness and time absorption, the eigen value can be determined. Effective cross sections can be weighted and calibrated with the calculated place- and energy-related flux.

Method

The Boltzmann equation is approximated by the S_N method. (The discrete S_N approximation to transport theory, LA - 2595, C.E. Lee, 1962). The scatter term is developed in 1st order Legendre polynomes, the place and angle coordinates are discretised, while the energy dependence is treated using the multigroup method.

Input

Selection of the geometry and order of the scatter, fixing of the machine grid for place and angle coordinates, specification of the S_N constants, allocation of the various materials to the place grid, selection of the effective cross sections from existing libraries or direct input of effective cross section assumptions.

5. Results

The results of the shielding calculations are given in Table 3 for a load of 33 LEU fuel assemblies, separately according to the gamma and neutron dose rate.

Area	Distance m	Dose rate in µ Sv/h - -	- n -	Total
Shell	0	770	83	853
	1	146	27	173
	2	55	16	71
Lid	0	795	6	801
	1	221	2	223
	2	94	1	95
Lid	0	182	1	183
	1	52	<1	52
	2	22	<1	22

Table 3: Results of the Shielding Calculations on the GNS 11 with a Load of 33 LEU Fuel Assemblies

The total results for a load of 33 LEU fuel assemblies are thus below the limits of the transport regulations as given at the beginning.

MTR fuel assemblies of medium initial enrichment (MEU F/As) with comparable burn-up are also covered with regard to shielding by the safety framework set by the LEU and HEU fuel assemblies.

For this reason, the limits of the transport regulations are once again observed when the casks are loaded with 33 MEU fuel assemblies and with mixed loads of GNS 11 including HEU, MEU and LEU fuel assemblies.

References

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/2/ R.G. Jaeger et. al.
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Springer Verlag 1968

/3/ ESIS Newsletter 34, p. 9 Table 9
Gamma-Ray-Flux-to-Dose-Rate Conversion Factors, July 1980

/4/ ZYLIND
An Interactive Point Kernel Program for Photon Dose Rate Prediction of Cylindrical Source/Shield Arrangements
NEA 1251/01

/5/ ANISN, Computer Code Collection, CCC -254
Oak Ridge National Laboratory (1976)